Westinghouse Non-Proprietary Class 3

WCAP-15996-NP
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Technical Description Manual for the CENTS Code

Volume 1

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TECHNICAL DESCRIPTION MANUAL FOR THE CENTS CODE

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ABSTRACT

CENTS is an interactive computer code for simulation of the Nuclear Steam Supply System and related systems. It calculates the behavior of a PWR for normal and abnormal conditions including accidents. It is a flexible tool for PWR analysis which gives the user complete control over the simulation through convenient input and output options.

CENTS is an adaptation of design computer codes to provide PWR simulation capabilities. It is based on detailed first-principles models for single and two-phase fluids. Use of nonequilibrium, nonhomogeneous models allows a full range of fluid conditions to be represented, including forced circulation, natural circulation, extensive coolant voiding and lower mode operation. The code provides a comprehensive set of interactions between the analyst, the reactor control systems and the reactor. This allows simulation of multiple failures and the effects of correct and incorrect operator actions. Examples of simulation runs with CENTS are steady state, power change, pump trip, loss of load, loss of feedwater, steam line break, feedwater line break, steam generator tube rupture, anticipated transients without scram, rod ejection, loss of coolant accidents, anticipated operational transients, and malfunctions of components, control systems or portions of control systems.

The CENTS code models most of the nuclear steam supply system and related systems. Core power is computed using a point kinetics model. Boiling curves for forced convection and pool boiling are used in the multi-node core heat transfer model. Primary and secondary thermal-hydraulic behavior is calculated with detailed multi-node and flowpath models. Nonhomogeneous, nonequilibrium conditions are also modeled, as well as the transport of solutes and non-condensible gases. The main control systems for reactivity, level, pressure, and steam flow are simulated. A multi-node and flowpath representation of the feedwater system is provided. Related balance of plant systems for single-phase fluid are represented.

The code features a Generic Control System design that processes system parameters and produces signals to drive the various plant subsystems. The control system is modular in design. It is constructed by the system modeler to simulate a specific plant's control systems, and can be made as simple or as true to the actual controllers as desired. It has an inventory of predefined functional modules, including arithmetic, Boolean, integro-differential and specialized functions. Once the control system structure is established by the modeler, it functions automatically, and its details do not normally concern the CENTS user. However, the user can, at any time during a transient, interactively change setpoints, disable control systems or exercise manual control. The control system designer/analyst, on the other hand, may study the detailed functioning of control modules by tracing their dynamic behavior, experimenting with their parameters and algorithms, or interfering with the lines of communication among the control modules.

The CENTS database provides a complete description of the nuclear plant systems that are modeled. Multiple plant states can be maintained on disk simultaneously as independent "snapshot" files, each of which contains the database plus a complete set of transient information. To initiate a transient from any snapshot, the user effects appropriate changes or perturbations to the plant. A new plant state is obtained by running a simulation to maneuver the plant interactively from a given plant state, or by using the code's self-initialization feature. Any intermediate state during a transient simulation can be saved as a snapshot for later study or to initiate parametric variations on plant behavior.

Use of CENTS is supported by executive software that handles most of the simulation mechanics, and allows the user to interact with CENTS as the transient progresses. This software provides a sophisticated command language that supports basic simulation maneuvers, collection of transient data, as well as complex interactions and probes by the advanced user. The executive software supports changes desired in the course of the transient and facilitates evaluation of the plant behavior details. The user may freeze, resume or backtrack a transient simulation at any time, examine plant parameters, make changes, take manual actions and initiate malfunctions. The user may, at any time, instruct the code to automatically make changes, display parameters or take any interactive action at pre-selected times, or when pre-selected dynamic conditions are satisfied, if desired, without interrupting the simulation.

CENTS may also be driven by a graphical environment in which the user accesses an interactive menu system via mouse-driven controls. Its live system parameter plots and graphical depictions of the plant state are invaluable tools in helping the user gain an understanding of the system behavior. In addition, a combination of standard and user-defined numerical outputs allows the user to explore details of the plant subsystem behavior.

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1.0 INTRODUCTION

CENTS is an interactive, high fidelity computer code for simulation of the Nuclear Steam Supply System (NSSS) components shown in Figure 1.1. It calculates the transient behavior of a PWR for normal and abnormal conditions including accidents. CENTS determines the core power and heat transfer throughout the NSSS. It also computes the thermal and hydraulic behavior of the reactor coolant in the primary and secondary systems. It includes the primary and secondary control systems and the balance-of- plant fluid systems.

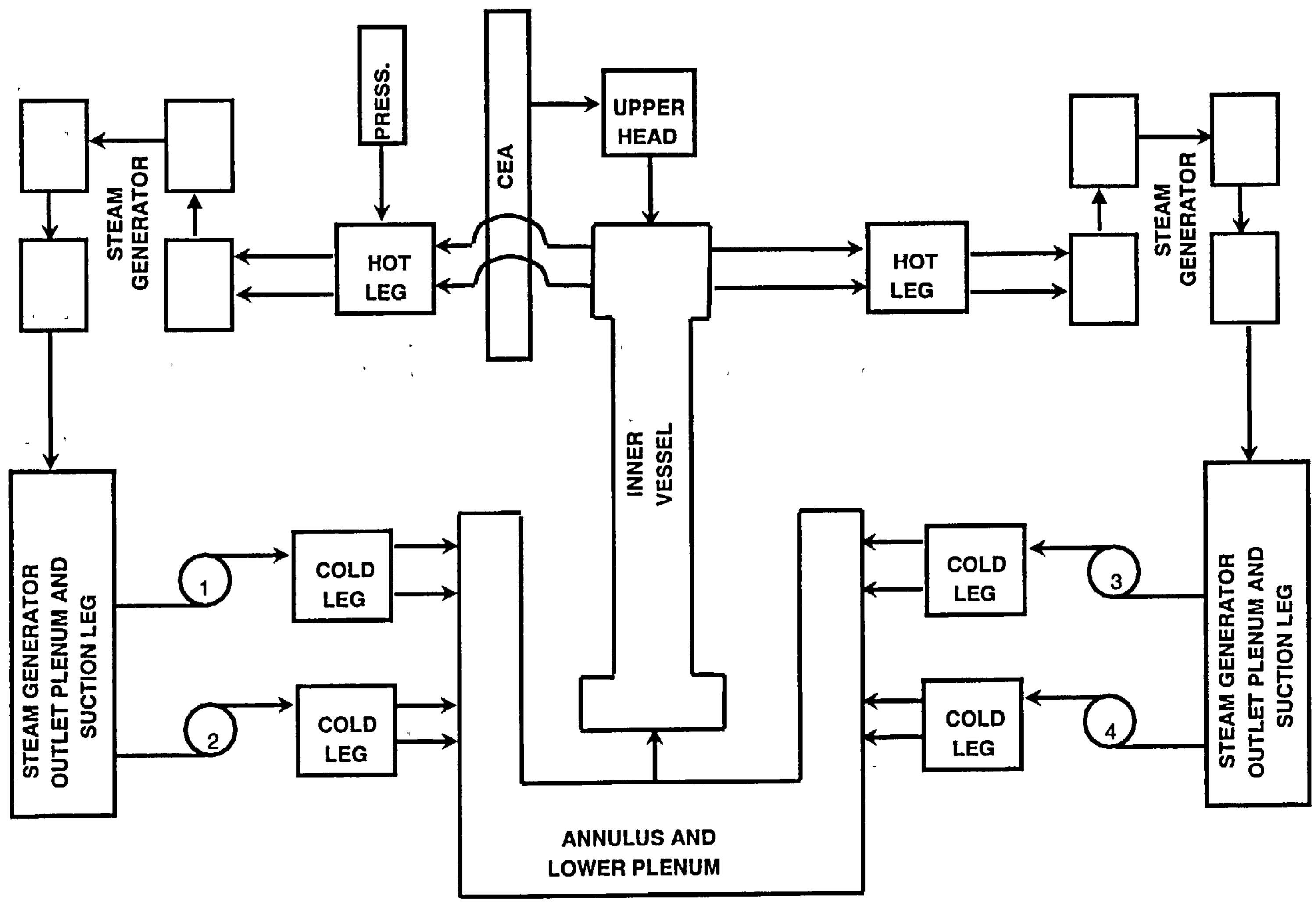
CENTS incorporates a number of features that enhance its usefulness. First-principles models provide a high degree of fidelity and flexibility. The database provides a complete set of input, and the plant state for full power operation. This reduces the input requirements to the user to those needed to initiate the desired changes in the plant state. The executive software provides a user-friendly command driven or a graphical user interface to control the code and the transient. Restart capability is provided. Output is available in several forms - summary prints at user selected intervals, brief or detailed edits constructed by the user, live parameter plots, and a live graphical display of the plant condition. The code runs faster than real time, which allows the user to gain a sense of the real-time plant behavior.

CENTS is designed to support engineering, operations, and training functions. It supports evaluation of plant behavior for accidents, for operator actions, for design, or for scoping studies. It is licensed for Chapter 15 (non-LOCA) safety analyses of pressurized water reactors designed by CE and Westinghouse. In addition, it may be used for NSSS simulations supporting optimization, procedure preparation or evaluation, determining success paths for PRA events, training, etc. The code simulates a wide range of variations in plant state, from steady state conditions to severe accidents. It provides a full range of interactions between the analyst, the reactor control systems and the NSSS. It also allows analysis of multiple failures and the effects of operator intervention or mistakes. Examples of transients run with CENTS include: steady state, power change, pump trip, overcooling or undercooling, loss of feedwater, steamline break, feedwater line break, steam generator tube rupture, anticipated transient without scram, loss of coolant accident, letdown line break, control rod ejection, and malfunctions of the various plant components and control systems. In addition, CENTS models lower mode operation with the NSSS either sealed or open to the containment/atmosphere, and abnormal events from lower mode, such as the loss of shutdown cooling (residual heat removal) from mid-loop operation.

This manual describes the CENTS models, the input variables and the output variables. Chapter 2 provides a brief summary of the models. Chapter 3 describes the core power and heat transfer models. Chapter 4 discusses the Reactor Coolant System (RCS) models, including the thermal-hydraulics and transport models, the primary loops, pressurizer, reactor coolant pumps, and quench (pressure relief) tank. Chapter 5 describes the secondary system models including the steam generator tubes, steam generators, main steamline and header, and feedwater systems. Chapter 6 presents the control and balance of plant models, user control of the related systems, and data base variables for the related control systems. Included are the following systems - reactor protective, pressurizer level and pressure control, chemical and volume control, relief valves, safety injection, turbine control, steam dump and bypass, feedwater, and control rods. Chapter 7 describes the input and output variables for the core, primary system, secondary system, malfunctions and accidents. It also explains how to access the dynamic data for changing parameters or obtaining details of the plant behavior. Chapter 8 provides the original verification studies for CENTS. Detailed studies for Chapter 15 events appear in Volume III. Several appendices provide additional detail about the plant definition and initial state input, tabular data, the generic controller system, models for CESEC emulation, and the CENTS link to CETOP.

Figure 1.1

TYPICAL CENTS MODEL OF TWO-LOOP PRESSURIZED WATER REACTOR



2.0 SUMMARY OF METHODOLOGY

CENTS is designed to provide realistic thermal-hydraulic responses for a full range of reactor operating conditions. The models are based on those found in computer codes used for PWR design. The core and primary system or Reactor Coolant System (RCS) models are based on those used in a best estimate, design version of the CEFLASH-4AS computer code[2.1]. This code provides realistic RCS responses for a full range of system conditions including a loss-of-coolant accident. The secondary system models are based on the Long Term Cooling (LTC) computer code[2.2]. Modeling capabilities are provided for control systems and balance of plant components. The code is used to evaluate the integrated plant response to operational and accident conditions. Highlights of the more important models are given here.

2.1 Primary System Geometric Representation

A node and flowpath network models the primary system thermal-hydraulic response. The nodes enclose control volumes, which represent the fluid mass and energy. Flowpaths connecting the nodes represent the fluid momentum and have no volume. The separation of mass and energy into control volumes and momentum into flowpaths is similar in concept to that used in the FLASH-4 code[2.3].

Figure 2.1 shows a typical geometric representation of the primary system of a PWR. Nodes are typically provided for the major components in the reactor primary system - inner vessel, upper head, control element assembly guidetubes, hot legs (including the associated steam generator inlet plenum), a pressurizer, steam generators (separate hot and cold sides of the U-tubes), combined outlet plena and suction cold legs, discharge cold legs and reactor coolant pumps, and the reactor vessel downcomer. The separate cold leg representations provide appropriate responses for partial loop operation. The interface with the secondary system occurs at the steam generator tube bundles.

Flowpaths with momentum solutions connect the nodes in the RCS model. [ ] In addition, non-momentum flowpaths are provided for addition and removal of coolant from the [ ]

The noding and flowpath modeling detail is not hardwired. The detail in representation of the system can be changed through input during the modeling process to tailor the representation to the requirements of a particular PWR design or event. Geometric features such as the number of loops or the number of steam generators as well as geometric details such as node volume, elevations, piping resistance, etc., are specified in a data base. Thus, adaptation of the RCS models for various plant layouts does not require recoding of the software.

2.2 Conservation Equations

The RCS thermal-hydraulic model is formulated with five one-dimensional conservation equations. The conservation variables are mixture (liquid and steam) mass, liquid mass, mixture energy, steam energy, and mixture momentum. The mass and energy for the liquid and steam are calculated for each node. Mass flowrate is calculated for each flowpath. The code incorporates slip effects in the flowpaths by means of [ ]

The conservation equations are integrated implicitly by means of a simultaneous solution of the linearized, discretized conservation equations. This yields a (4xM+N) by (4xM+N) system of linear equations, where M is the number of nodes and N is the number of momentum flowpaths. The structure of the coefficient matrix permits use of a system reduction procedure that produces a NxN system of equations. The NxN matrix is solved using a block inversion technique[2.5]. After solution of the conservation equations, the pressure in each node is calculated. The mass and energy in the liquid and steam regions are used with the water property correlations and nodal volume to determine the resulting state variables (pressure, phase enthalpies and phase temperatures).

2.3 Two-Phase Fluid Representation

CENTS models phase separation within a node into a separate steam region and a liquid or two-phase region consisting of a continuous liquid phase with dispersed bubbles. Bubbles are generated by heat sources within the node, flashing in the node or transport from the adjacent nodes. Phase separation is calculated in terms of bubble rise velocities that are found from [ ] A more detailed model provides an axial void fraction distribution in the inner vessel node[2.7]. Nodes with phase separation provide a discrete two-phase mixture level in the node. Where appropriate, the fluid level impacts on heat transfer rates and on the quality of fluid exiting through flowpaths connected to the node.

2.4 Thermal Nonequilibrium States

CENTS provides a full range of thermodynamic fluid states for all primary nodes. Nodes with homogeneous or fully mixed fluid are at thermal equilibrium. The possible thermal nonequilibrium states for non-homogeneous or phase-separated nodes with separate two-phase mixture and steam regions are (a) saturated liquid with saturated steam (equilibrium), (b) subcooled liquid with saturated steam, (c) saturated liquid with superheated steam, and (d) subcooled liquid with superheated steam. The model dynamically calculates the node thermodynamic state from one of these states. It includes a detailed flow regime dependent condensation model that considers condensation of bubbles, vapor condensation on an injected subcooled liquid, condensation at the surface of a liquid pool, vaporization due to wall heat, and the effect of non-condensible gases. In addition, energy partitioning of wall heat transfer between the liquid (two-phase) and steam regions is calculated.

2.5 Reactor Core

A point kinetics model calculates reactor power. The point kinetics model distributes the calculated power according to a user-supplied axial power shape. Axially varying heat transfer and core coolant data is provided to this model to support a full range of temperature and moderator driven feedbacks.

The core heat transfer model covers the full range of fluid conditions in the core. Provision is made for forced convection and quiescent pool boiling conditions. The mode of heat transfer is determined dynamically. Boiling curves for both conditions include subcooled, nucleate boiling, transition boiling, film boiling, and steam heat transfer correlations. Fuel temperature, cladding temperature, and the heat flux (hence, the heat transfer regime and surface coefficients) are calculated implicitly. Appropriate radial and axial noding detail is used in the fuel rods.

2.6 Other Primary System Models

CENTS models all primary system components. Some additional models of interest are discussed here. Reactor coolant pump (RCP) performance is modeled via the equation for conservation of angular momentum, along with RCP homologous curves for head and torque in single- and two-phase conditions, with two-phase head degradation. The code models wall heat transfer, pressurizer heaters, pump heat, and core bypass flow. It calculates coolant inventory changes due to the operation of systems such as the emergency core cooling system, the chemical and volume control system, and auxiliary pressurizer spray. Critical (choked) flow out of the primary system, through leaks or relief valves, is found from [

] The code provides models for injection or generation, transport and inter-phase transfer of solutes including non-condensible gases. Coolant thermodynamic properties are given by a set of efficient water property correlations that provide continuous property derivatives for a full range of fluid conditions -

subcooled, saturated, and superheated. A reactor vessel mixing model determines the tilt in the inlet temperatures of the hot legs that results from asymmetric operation of the steam generators.

2.7 Steam Generator Heat Transfer

The steam generator model performs separate heat transfer calculations for the hot and cold sides of the tubes. The hot and cold sides of each steam generator's tubes may each be represented by one or two primary nodes. Each of these nodes may be further segmented for detailed representation of the heat transfer and temperature profiles.

Models for forward and reverse heat transfer are incorporated in the steam generator heat transfer logic. The overall heat transfer is determined from film resistance of the primary and secondary sides and the wall resistance. The primary side film resistance for forward heat transfer is found from correlations for subcooled forced convection and two-phase flow with condensation. The primary side reverse heat transfer is found from correlations for nucleate boiling and heat transfer to steam. The secondary side film resistance is calculated using correlations for pool boiling and heat transfer to steam. The coolant levels and their effect on the fluid state, heat transfer area and heat flux are modeled on both the primary and secondary sides of each steam generator.

2.8 Steam Generator Secondary System Geometric Modeling

Three nodes are typically used for the secondary side of each steam generator - a downcomer (saturated or subcooled), an evaporator region (saturated, with a subcooled layer), and a steam dome (saturated or superheated) as shown by Figure 2.2. The common steamline header is represented by one or two additional nodes. This system representation allows accurate modeling of the recirculation phenomena and the downcomer and evaporator water levels. All major components are modeled, including the secondary safety valves, atmospheric and condenser dump valves, steam isolation valves, turbine bypass and turbine admission valves, and main and auxiliary feedwater systems.

2.9 Secondary System Conservation Equations

The code enforces mass and energy balances for each node. The circulation flow and the main steam flows are calculated using momentum balances. The steam flow from the evaporator to the steam dome is calculated using a [ ] model. The pressure and remaining state properties are calculated from the mass and energy in each node. Flowrates within, into and out of the steamline header are represented by a coupled system of equations that accounts for choked flow conditions in the steamline, and for the effect of steamline check valves if they are present.

2.10 Other Secondary System Models

Models are provided for the steamlines, steamline header(s), and several types of valves including main steam isolation, turbine admission, turbine bypass, atmospheric dump valves, and secondary safety valves, etc. The resulting equations representing these systems are solved simultaneously to calculate flowrates and pressures in each steamline and header.

Transport, release and partitioning of solutes including iodine is modeled in each steam generator node, steamline and header. Direct iodine release to the atmosphere as well as leakage from the containment, condenser and the turbine is modeled. The resulting iodine release is used in a dose model.

A detailed main and auxiliary feedwater system model is provided. It provides component models for leaks and breaks in the piping, several types of valves and pumps, the feedwater heaters, the condenser, the heater drain tanks, and the feedwater piping cross connects. The feedwater system configuration is defined by input, with N nodes to conserve mass, and M connecting flowpaths with a steady-state momentum equation for each pipe (path).

2.11 Control System Modeling

CENTS features a flexible, modular method to handle control systems for the core, primary system, secondary system and selected ancillary systems. Interfaces between the controllers, ancillary systems, process models and user are provided. The controllers are built by the system modeler (through inputs) from modules that perform the standard arithmetic, integro-differential and logical transforms used in control circuits. This provides a generic capability to handle plant specific characteristics of control systems and operator actions.

2.12 Balance of Plant Systems

Component models for the following ancillary systems are provided to represent the balance of plant systems: reactor protective (scram) system, pressurizer level and pressure control systems, chemical and volume control system, relief valves, safety injection system, turbine control and steam regulating systems, feedwater systems, and control rod regulating system. These are linked to the controllers for manual or automatic operation. Models to initiate and support the usual operational and accident events (malfunctions) including small break LOCA and letdown line break, reactor coolant pump failures, valve failures, steam generator tube rupture, steamline break, and feedwater line break are provided.

2.13 References

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2.6 "Calculative Methods for the C-E Small Break LOCA Evaluation Model", CENPD-137, Supplement 1 (Non-proprietary), January 1977.

2.7 T. M. Anklam and M. D. White, "Experimental Investigations of Two-Phase Mixture Level Swell and Axial Void Fraction Distribution Under High Pressure, Low Heat Flux Conditions in Rod Bundle Geometry", pp. 4-67 to 4-88, ANS Specialists Meeting on Small Break Loss of Coolant Accident Analyses in LWRs, Monterey, California, EPRI WS-81-201, August 25-27, 1981.

2.8 F. J. Moody, "Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels", ASME Winter Annual Meeting, November 1975.

2.9 R. E. Henry and H. K. Fauske, "The Two-Phase Critical Flow of One-Component Mixture in Nozzles, Orifices and Short Tubes," Journal of Heat Transfer, May 1971.

Figure 2.1
TYPICAL PRIMARY SYSTEM GEOMETRIC MODEL
(3-LOOP W PWR)

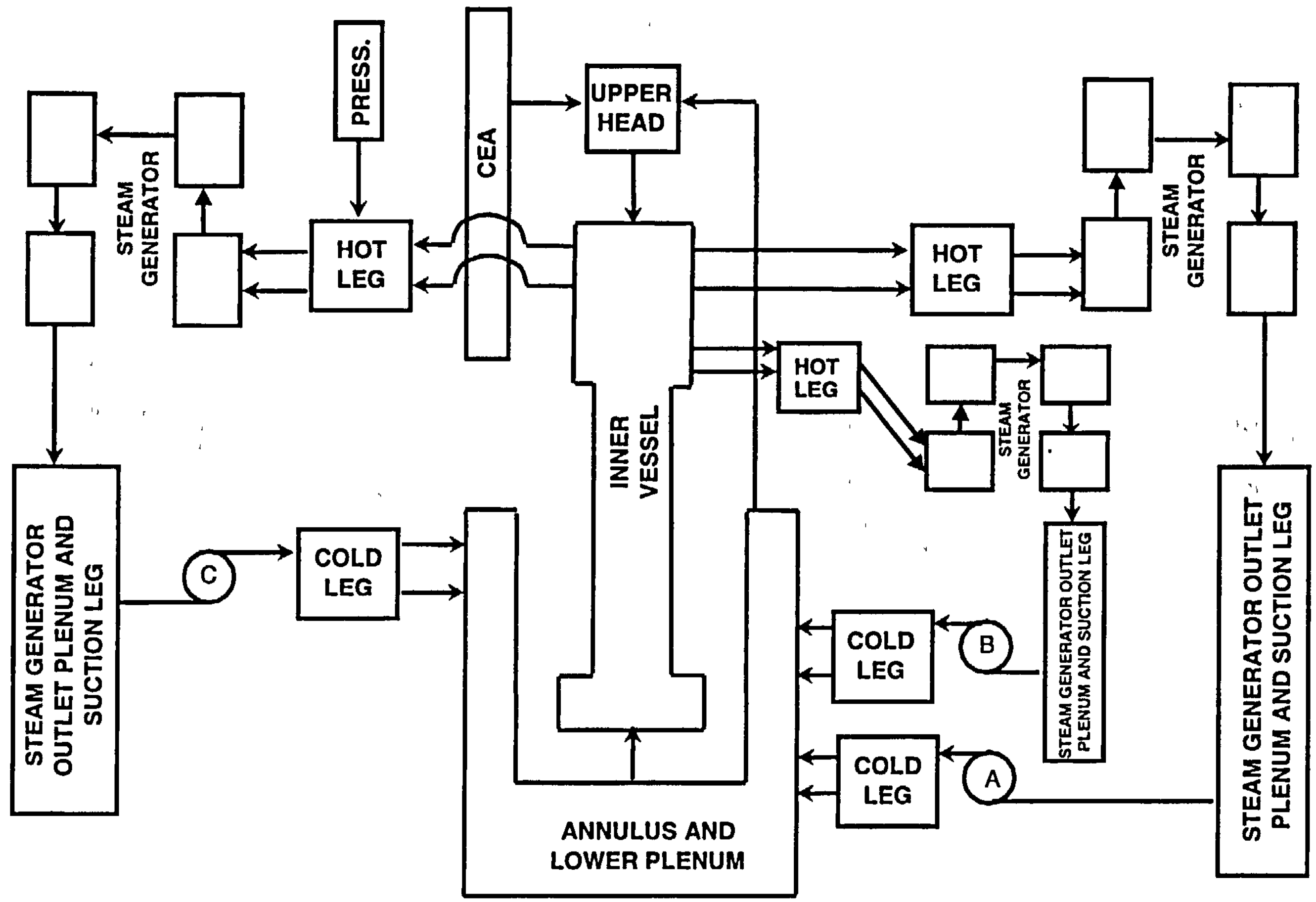
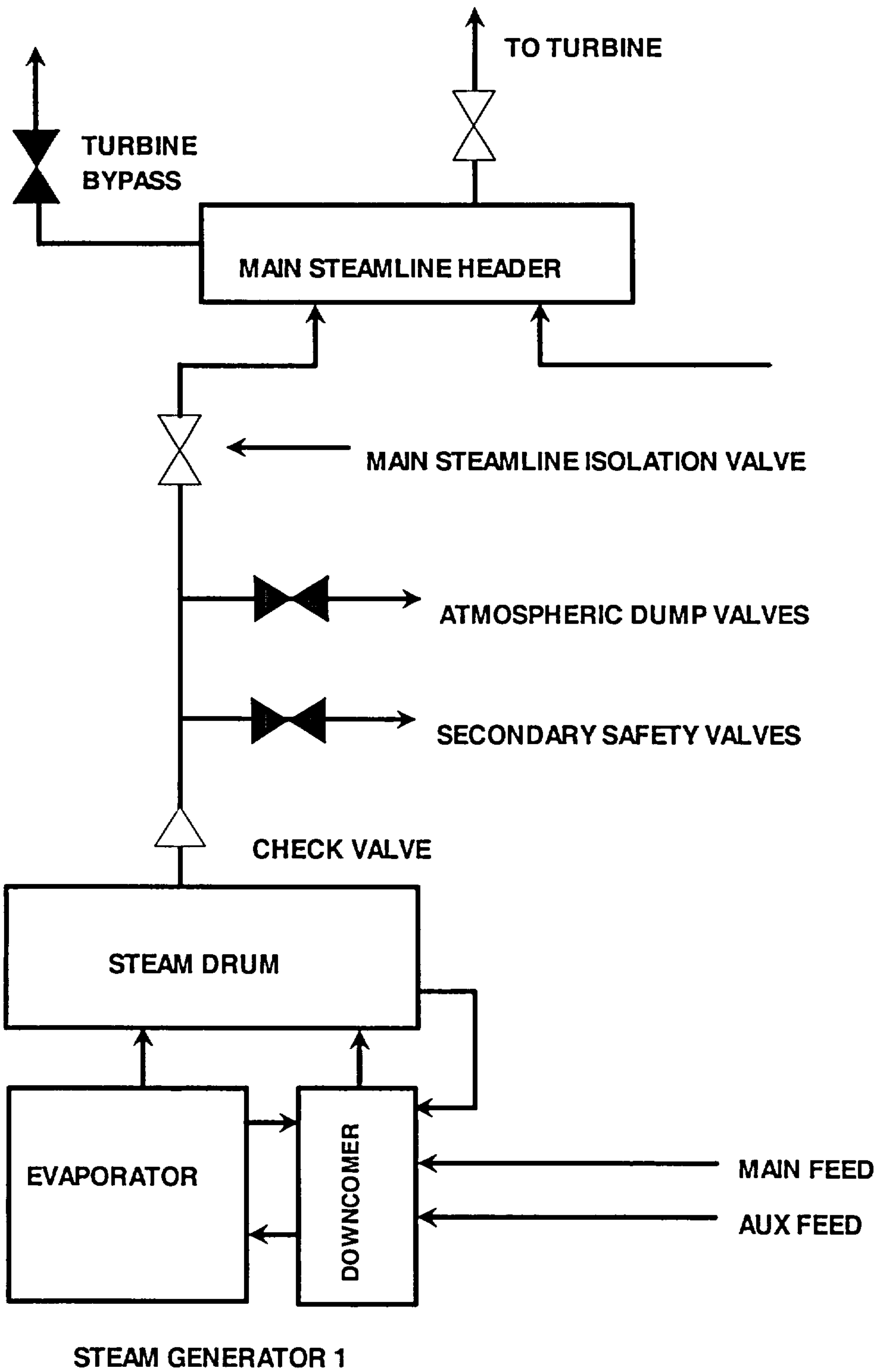


Figure 2.2

TYPICAL STEAM GENERATOR SECONDARY SYSTEM GEOMETRIC MODEL



3.0 CORE MODEL

The core model includes the core power, oxidation of zirconium cladding, and core heat transfer. .

3.1 Core Power by Point Kinetics

The core power model describes all of the nominal power sources in a reactor – fission power, decay power and zirconium-water reaction. Point kinetics is used to describe the time dependent response of core [

] until it is no longer appropriate. A user specified power history may also be imposed. An input axial power shape is combined with the kinetics, decay heat, and user power results to provide an appropriate power distribution using the same axial nodalization as the core heat transfer solution.

3.1.1 Point Kinetics Model

The point reactor kinetics and decay heat equations of FLASH[3 16] are used with minor modifications including addition of a fixed source term and use of an improved solution methodology for the point kinetics equations. The fission power input to the fuel is found from the reactor kinetic equations with six delayed groups. The decay power is based on the fission product inventory that would result from long term steady-state operation at the specified initial power level. Optional kinetics input includes provision for changing the effective delayed neutron fractions and decay constants and using non-equilibrium initial concentrations. Optional decay heat input includes provision for changing the fractional power, decay constants, and initial concentrations of the eleven fission product groups.

Initial conditions are set up by assuming that the reactor has been operated at the initial power level for an infinite time. Scram can be initiated by the user or by the automatic trip system described in Section 6.1.

The total time varying power output is

$$P(t) = n(t)\, E_I + \sum_{j=1}^{11} \lambda_{Dj} E_{Dj} C_{Dj}(t) \tag{3.1}$$

where

$P(t)$ = core power output at time t
$n(t)$ = total fission rate in the core at time t
E_I = prompt fission energy release
λ_{Dj} = decay constant of fission product j
E_{Dj} = decay energy of fission product j
$C_{Dj}(t)$ = concentration of fission product j at time t

The first and second term on the right hand side of Equation (3.1) are instantaneous power and decay heat respectively.

The production of fission products is governed by

$$\frac{d}{dt} C_{Dj}(t) = \gamma_{Dj}\, n(t) - \lambda_{Dj}\, C_{Dj}(t) \tag{3.2}$$

where γ_{Dj} = yield rate of fission product j.

The above two equations become for a steady state

$$P(0) = n(0)\,E_I + \sum_{j=1}^{11} \lambda_{Dj} E_{Dj} \tag{3.3}$$

$$C_{Dj}(0) = \gamma_{Dj}\, n(0) / \lambda_{Dj} \tag{3.4}$$

Thus a normalized power output is calculated by

$$\frac{P(t)}{P(0)} = \frac{n(t)}{n(0)} \frac{E_I}{Y} + \sum_{j=1}^{11} \frac{\gamma_{Dj}\, E_{Dj}\, \lambda_{Dj}\, C_{Dj}(t)}{Y\, \gamma_{Dj}\, n(0)} \tag{3.5}$$

where

$$Y = E_I + \sum \gamma_{Dj} E_{Dj}.$$

Introducing a normalized fission product concentration

$$\chi_{Dj} = \lambda_{Dj} C_{Dj}(t) / \left(\gamma_{Dj} n(0)\right)$$

simplifies the normalized power equation to be

$$\frac{P(t)}{P(0)} = \left(1 - \sigma_D\right) \frac{n(t)}{n(0)} + \sum_{j=1}^{11} \alpha_{Dj}\, \chi_{Dj} \tag{3.6}$$

where

$$\alpha_{D_j} = \frac{\gamma_{Dj} E_{Dj}}{Y}$$

$$\sigma_D = \sum_{j=1}^{11} \alpha_{Dj}.$$

The fission rate and delayed neutron precursor concentrations, C_{pi}, are evaluated using the conventional kinetics equations:

$$\frac{dn(t)}{dt} = \frac{\rho - \bar{\beta}}{\ell^*} n(t) + \sum_{i=1}^{6} \lambda_{pi} C_{pi}(t) + S_{neutron} \tag{3.7}$$

$$\frac{d}{dt} C_{pi}(t) = \frac{\bar{\beta}_i}{\ell^*} n(t) - \lambda_{pi} C_{pi}(t) \tag{3.8}$$

where

ℓ^* = prompt neutron lifetime

ρ = total reactivity (components described in Section 7.1.1)

λ_{pi} = decay constant of delayed neutron precursor group i

$C_{pi}(t)$ = concentration of delayed neutron precursor group i at time t

$\bar{\beta}_i$ = effective delayed neutron fraction for delayed neutron precursor group I

$\bar{\beta}$ = total beta fraction

$S_{neutron}$ = effective neutron source term

The equations are normalized by introducing the normalized precursor concentration

$$x_{pi}(t) = \frac{\ell^* \lambda_{pi} C_{pi}(t)}{\bar{\beta}_i \, n(0)}$$

the normalized source term

$$S = S_{neutron} \, \ell^* / n(0)$$

and the fractionalized core power

$$P(t) = n(t) / n(0)$$

The resulting equations are

$$\frac{d}{dt} P(t) = \frac{\rho - \bar{\beta}}{\ell^*} P(t) + \frac{1}{\ell^*} \sum_{i=1}^{6} \bar{\beta}_i \chi_{pi}(t) + \frac{S}{\ell *} \tag{3.9}$$

and

$$\frac{d}{dt} \chi_{pi}(t) = \lambda_{pi} \left(P(t) - \chi_{pi}(t) \right). \tag{3.10}$$

The following reactivity feedbacks are provided – moderator boric acid concentration, moderator temperature, Doppler (fuel temperature) and control rod motion as described in Section 7.1.1. Optionally, the moderator boric acid and temperature feedbacks may be replaced by a combined moderator density feedback. These effects are accounted for through tabular input.

The equations for computing the feedback reactivities are

$$\rho_y(t) = f(y) - f(y)^{t=0} \tag{3.11}$$

where

y = independent variable at time t, e.g., fuel temperature for Doppler effect,

$f(y)$ = feedback reactivity as a function of y,

$f(y)^{t=0}$ = feedback reactivity at beginning of transient.

The reactivity is calculated as the sum of control rod, fuel temperature (Doppler) and moderator contributions:

$$\rho = \sum \rho_y + \rho_o \tag{3.12}$$

where ρ_o is the total reactivity at the beginning of the transient.

The kinetics equations are solved using a central difference formula

$$y(\Delta t) = y(0) + \{ \dot{y}(0) + \dot{y}(\Delta t) \} \Delta t / 2$$

This approximation neglects terms of the Taylor expansion that are of order Δt^3 or higher. The point kinetics equations are integrated over a time interval that is the same as that used by other CENTS algorithms. Reactivity is constant over this interval. CENTS integrates the resulting equation by dividing this time interval into smaller Δt intervals and solving it for each subinterval as follows:

1. Choose the neutronics time step Δt. It is limited to the range $0.0005 \le \Delta t \le 0.05$ seconds.
2. If necessary reduce Δt such that the denominator of the equation is not zero or negative.

3. For each subinterval Δt, use a first order approximation for the initial value of χ_I and calculate $P(t + \Delta t)$. Then recalculate χ_I using a second order approximation and use it as the starting point for the next time step. If power has changed by more than 2% during the current kinetics time step, set

$$\Delta t = 0.9\ \Delta t\ (0.02 / \text{Power Change})$$

and restart the calculation at Step 1. This provides a much more restrictive constraint than Step 2.

This methodology increases the accuracy of the solution for large reactivity insertions and supports use of larger time steps.

The fission product decay parameters, α_{Dj} and λ_{Dj}, were taken from Reference 3.16. However, the delayed neutron precursor parameters, β_i and λ_{pi}, are specified for the plant state in question.

3.1.2 Decay Heat Curve

The revised ANS/ANSI 5.1 decay heat curve[3.17] has been incorporated into the CENTS data base. The correlations of the standard were evaluated for a typical PWR operating on a realistic fuel cycle at nominal conditions. No uncertainties (zero sigma) were incorporated. Heavy element decay (^{239}U and ^{239}Np) and fission product neutron capture are represented. The code is switched to the ANS decay heat curve at the time when the[

]

The decay heat curve is then used for the remainder of the transient. A multiplier can be applied to the decay heat curve to account for uncertainties and redistribution effects.

3.1.3 Zirconium-Water Reaction

The reaction rate for the zirconium-water reaction follows the parabolic relationship derived by Baker and Just[3.1]. For each time step the metal-oxide interface velocity is calculated from the parabolic rate equation assuming no change of temperature during this step. The solution is done for multiple axial nodes using the same nodalization used in the fuel rod temperature model. Once the metal-oxide interface velocity is known, the heat generation rate per unit area and mass of hydrogen release are calculated. The energy generated is deposited in the fuel cladding. The hydrogen generated is released into the reactor coolant in the core.

3.2 Other Core Models

3.2.1 Fuel Failure

The fuel failure model predicts the extent of fuel damage, the subsequent fission product release and the rate of hydrogen generation. The fuel failure threshold is based on three design limits: (1) the hot channel Departure from Nucleate Boiling Ratio (DNBR), (2) the peak cladding temperature and (3) the amount of clad oxidation.

Departure from nucleate boiling (DNB) is usually localized and of short duration, resulting in an increase in coolant activity whose magnitude depends on the number of pins experiencing DNB. High cladding temperatures, which can occur during periods of core uncovery, result in the oxidation of zirconium, generation of hydrogen and production of reaction heat, based on the Baker-Just correlation. Fuel pins that are oxidized beyond 17% or whose cladding temperature exceeds 2200°F are assumed to have failed. In some cases, the amount of fuel damage as a result of DNB may be defined as the product of the number of pins within a given pin power and the probability of DNB.

The fission product inventory available for release upon fuel failure is dependent on the power history. The model accounts for seven isotopes of iodine, xenon and cesium.

3.2.2 *DNBR Fuel Design Limit*

CENTS uses the same methods as are used with CESEC to determine the approach to fuel design limits. Approach to the DNBR fuel design limit is determined using the system response data calculated by CENTS (average heat flux and RCS pressure, temperature and flow) in combination with conservative axial and radial peaking factors. DNBR is calculated using the TORC or CETOP codes. The HRISE code is used for steamline break return-to-power conditions. The DNBR calculation in CENTS is included for information and to support the fuel failure model.

3.3 Core Heat Transfer

The Core Heat Transfer (CHT) model calculates the average rod temperature and the core average heat transfer to the coolant. The model calculates the heat transfer for an axially sectionalized rod based on local fluid conditions and local rod surface temperatures using complete boiling curves for the forced convection and pool boiling modes.

Basic features of the model are as follows:

1. The energy balance in the fuel rod is solved using a one-dimensional cylindrical volume averaged heat conduction equation.
2. The energy balance of the fluid is solved using a one-dimensional closed channel formulation.
3. Rod to coolant heat transfer is calculated using complete boiling curves for forced convection and pool boiling models.
4. The selection of forced convection or pool boiling modes is dynamically calculated.

5. The fuel rod and fluid channel are axially sectionalized into as many as 20 axial sections.
6. The fuel (up to 8 regions), gap and cladding regions are modeled explicitly in the radial direction for each axial node.
7. Fission power is generated directly in the fuel, gap, cladding and coolant regions.
8. The thermal conductivity and heat capacity of the fuel and cladding are dynamically calculated.

The following is a description of the models and the equations.

3.3.1 <u>System Representation</u>

The Core Heat Transfer model (CHT) axially sectionalizes the core as shown in Figures 3.1 and 3.2. The number of axial sections is variable, [ ]. The number of radial regions is variable, [ ]

The coolant channel is also sectionalized axially. The number of axial sections is equal to the number of axial sections used for the core [ ]. In addition, a core inlet section is also included as an additional section for the coolant channel calculations.

The code calculates the rod temperature distribution [ ] and surface heat flux for each of the axial sections. The heat generation in the core is an axial (radially averaged) power profile. The axial coolant enthalpy profile is calculated using boundary conditions supplied by the primary coolant system models described in Chapter 4 (pressure, fluid flow into the inner vessel, mixture level, subcooled liquid level, etc.) and the local heat transfer calculated by the core heat transfer model.

3.3.2 *Discretized Heat Conduction Equation*

For each axial region z', bounded by interfaces z' – 1 and z', the volume averaged, discretized, implicit in time, centered at z', heat conduction equation is

(3.13)

where

$\overline{T}_i$ = region z' volume averaged temperature

A_i = interface i surface area

$\overline{K}_i$ = interface i average conductance

$M_i C_i$ = region z' total heat capacity

Q_i = region i heat rate generation

Δt = time step size.

3.3.3 *Integration of the Heat Conduction Equation*

Application of Equation (3.13) for the fuel, gap, and cladding regions produces a tri-diagonal system of equations in fuel temperature, gap temperature, cladding temperature, and heat flux. In order to complete the system of equations, equations for the heat flux and surface temperature are defined as follows:

$$\phi = \frac{K}{\Delta r}\,(T_{clad} - T_{surf})$$

and (3.14)

$$\phi = hA\,(T_{surf} - T_{cool})$$

where

K = cladding thermal conductivity

h = surface heat transfer coefficient

A = surface area

T_{clad} = cladding temperature

T_{fuel} = fuel temperature (up to 8 radial regions)

T_{gap} = gap temperature

T_{surf} = cladding surface temperature

T_{cool} = local average coolant temperature

Δr = cladding thickness

ϕ = heat flux.

Solution of the above system of equations yields new values for T_{fuel}, T_{gap}, T_{clad}, ϕ, and T_{surf}.

3.3.4 Heat Transfer Modes

Boiling curves for the forced convection and pool boiling modes are implemented. The forced convection boiling curve is applied[ ] It is always used during normal operation of the system. The pool boiling curve is used when the system attains a quiescent pool boiling mode of heat transfer. This mode of heat transfer usually occurs only during severe accidents including those leading to core uncovery. The code dynamically selects the forced convection or pool boiling mode.

Forced Convection Boiling Curve The forced convection boiling curve consists of the following regimes and correlations.

1.	Subcooled forced convection:	Dittus-Boelter(3 2)
2.	Nucleate boiling:	Thom(3 3)
3.	Transition boiling:	McDonough, Milich and King(3 4)
4.	Stable film boiling:	Dougall-Rohsenow(3 5)
5.	Heat transfer to steam:	Dittus-Boelter(3 2)
6.	Critical heat flux:	W-3(3 6) and (3 7) Macbeth correlations (3 7) and (3 8)

Pool Boiling Curve The pool boiling curve consists of the following regimes and correlations:

Subcooled natural convection[ ]

2.	Nucleate boiling:	Rohsenow(3.9)
3.	Transition boiling:	McDonough, Milich and King(3 4)
4.	Stable film boiling:	Modified Bromley(3.10)
5.	Heat transfer to steam:	Dittus-Boelter(3 2) and Sieder Tate(3.11)
6.	Critical heat flux:	Zuber(3.12)

Description of the Heat Transfer Coefficient Correlations The nomenclature and units for the heat transfer correlations are as follows:

h = heat transfer coefficient (Btu/ft^2 hr °F)

ϕ = heat flux (Btu/ft^2 hr)

W = mass flowrate (lbm/sec)

T = fluid temperature (°F)

T_{sat} = saturation temperature (°F)

T_s = surface temperature (°F)

P = pressure (psia)

A = flow area (ft^2)

D = hydraulic diameter (ft)

K = fluid conductivity (Btu/hr ft °F)

Re = Reynolds number

Pr = Prandtl number

X = quality

H_{in} = inlet enthalpy (Btu/lbm)

h_f = saturation liquid enthalpy (Btu/lbm)

h_g = saturation steam enthalpy (Btu/lbm)

ρ_f, ρ_ℓ = liquid density (lbm/ft^3)

ρ_g, ρ_v = steam density (lbm/ft^3)

C_p = specific heat (Btu/ft^3 °F)

μ = viscosity (lbm/ft-sec)

σ = Surface tension (lbf/ft)

L = channel length (ft)

Z_m = two-phase mixture height (ft)

The subscripts used are as follows:

ℓ, f = liquid phase

v, g = steam phase

DNB = departure from nucleate boiling

1. Subcooled Forced Convection and Forced Convection to Steam

   Dittus-Boelter

   $h = 0.023\ (K/D)\ Re^{0.8}\ Pr^{0.4}$

2. Nucleate Boiling – Thom

   $\phi = (\exp(2P/1260)/\ 0.072^2)\ (T_s - T_{sat})^2$

3. Transition Boiling – McDonough, Milich and King

   $\phi_{DNB} - \phi = \{730 \exp(576/P)\}\ (T_s - T_{DNB})$

4. Stable Film Boiling – Dougall-Rohsenow

   $h = 0.023\ (K_v/K)\ Re^{0.8}\ Pr_v^{0.4}$

5. Critical Heat Flux – W-3

   $\phi_{DNB} = C_1 C_2 C_3 C_4$

where

$C_1 = 2.022 - 0.0004302P + (0.1722 - 0.0000984P) * \exp\{(18.177 - 0.004129P)\ X\}$

$C_2 = (0.1484 - 1.596X + 0.1729X^2)\ W/A + 1.037 \times 10^6$

$C_3 = (1.157 - 0.869X)\ (0.2664 + 0.8357 \exp(-3.151 * 12D)$

$C_4 = 0.8258 + 0.000794\ (h_f - h_{in})$

The W-3 correlation is used if

a. $1000 < P < 2400$

b. $1000/3.6 < W/A < 5000/3.6$

c. $X < 0.15$

These conditions are usually encountered during normal operation.

6. <u>Critical Heat Flux</u> – MacBeth

$$\phi_1 = 6330\,(h_g - h_f)\,/\,(12D)^{0.1}\,\{W/(10^6 A)\}^{0.51}\,(1 - X)$$

$$\phi = \max\,(\phi_1, \phi_2)$$

The critical heat flux using the MacBeth correlation is calculated for conditions outside the range of the W-3 correlation.

7. <u>Nucleate Boiling</u> –[

]

8. <u>Stable Film Boiling</u> –[

].

9. <u>Heat Transfer to Steam</u> –[

]

[ ]

10. <u>Critical Heat Flux</u> –[

]

<u>Heat Transfer Logic</u> First, the code determines applicability of the forced convection or pool boiling correlations depending on the fluid conditions as described at the beginning of Section 3.3.4. Once the mode is selected, the code uses the correlations from the forced convection boiling curve or those from the pool boiling curve in an identical manner as follows:

[

]

[ ]

3.3.5 *Coolant Enthalpy Profile*

CENTS calculates the coolant enthalpy profile for the core and lower plenum for single-phase conditions (subcooled water or superheated steam), by one of two models, selected through the user input option, CHT_USE_NEW_ENTHALPY_OPTION.

Original Channel Enthalpy Model (CHT_USE_NEW_ENTHALPY_OPTION = 0.0).

In the original model, the coolant enthalpy profile is obtained by an explicit solution of the fluid energy equation for a closed channel:

$$W \frac{dh}{dz} = Q' \tag{3.15}$$

where W = mass flowrate
h = specific enthalpy
z = elevation
Q' = linear heat rate.

The mass flowrate used in the subcooled region is the mass flowrate entering the core. For the steam region, the mass flowrate is the steam flow leaving the two-phase mixture region. The linear heat rate, Q', includes both the rod surface heat transfer and the direct deposition of the core heat in the coolant.

This model ignores the heat capacity of the liquid, and assumes that the axial core enthalpy distribution follows the heat flux distribution with no lag. That is, the enthalpy of each core section is calculated from Equation (3.15) as

$$h_{j+1}(t) = h_j(t) + \frac{Q_{j+1}}{W_{core}} \tag{3.16}$$

where
- $h_j(t)$ = enthalpy in section j at time t
- $h_{j+1}(t)$ = enthalpy in section j+1 at time t
- Q_{j+1} = heat rate for section j+1
- M_{j+1} = mass in section j+1
- W_{core} = core flowrate (assumed to be the same for each section)
- Δt = calculational time step length

The above formulation is appropriate when the enthalpy of the section is changing slowly relative to the time constant $\tau = M / W$.

For the two-phase region (region bounded by the saturation liquid level, and the two-phase mixture level), the local two-phase enthalpy is calculated by the expression

$$h_{2\phi} = h_f + x\, h_{fg}$$

where x is the local quality calculated by the RCS inner vessel bubble disengagement model.

Improved Channel Enthalpy Model (CHT_USE_NEW_ENTHALPY_OPTION = 1.0).

This is an optional model feature. The rate of change of enthalpy in a core section that considers the heat capacity of the liquid is found from the differential equation

$$\frac{\delta}{\delta t} h_{j+1} = \frac{Q_{j+1} - (h_{j+1} - h_j) W_{core}}{M_{j+1}}$$

As in the original model, core flow is assumed to be the same for each of the core sections. M_{j+1} is based on the section enthalpy and node-average pressure. The equation for $h_{j+1}(t)$ is integrated analytically over Δt to avoid any instability that might occur with a finite-difference solution for large time steps. This yields:

$$h_{j+1}(t + \Delta t) = h_j(t) + \frac{Q_{j+1}}{W_{core}} + e^{-\frac{W_{core}}{M_{j+1}} \Delta t} \left(h_{j+1}(t) - h_j(t) - \frac{Q_{j+1}}{W_{core}} \right)$$

3.3.6 Fuel Rod and Fluid Properties

The fuel and cladding temperature-dependent thermal conductivities and volumetric heat capacities are calculated using polynomial fits described in References 3.13, 3.14, and 3.15. The gap thermal conductivity and volumetric heat capacity are constant. The CHT has an efficient, optimized calculation of fluid and material properties. The code completely updates the properties every time step.

3.4 References

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Figure 3.1

FUEL ROD AND COOLANT HEAT TRANSFER NODALIZATION

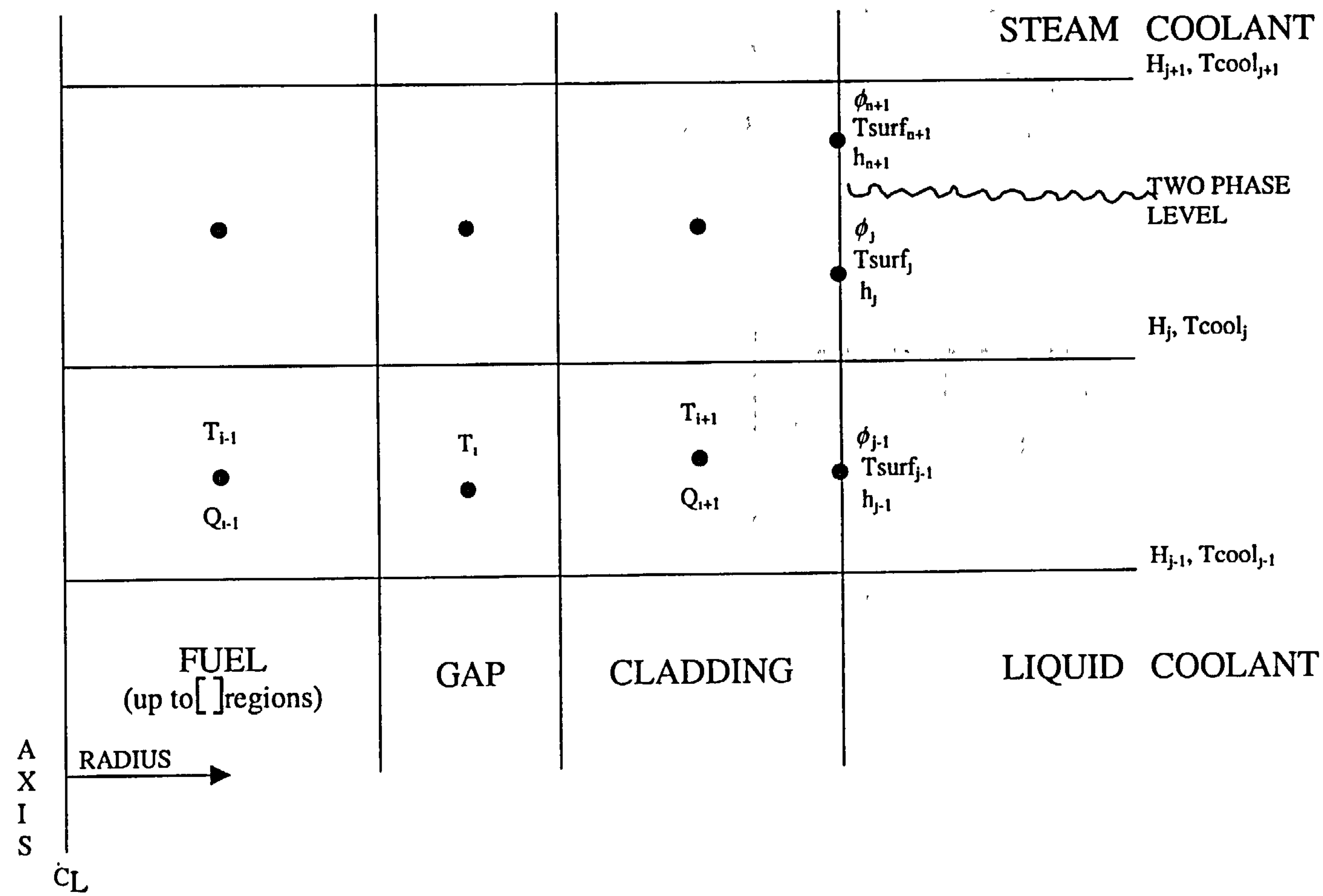

Figure 3.2

CORE AND CORE INLET REPRESENTATION